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## Standard Practice for
# CONDUCTING SURVEILLANCE TESTS FOR LIGHT-WATER COOLED NUCLEAR POWER REACTOR VESSELS[1]

This standard is issued under the fixed designation E 185; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon ($\epsilon$) indicates an editorial change since the last revision or reapproval.

$^{\epsilon}$ NOTE—Section 9.2.3 was corrected editorially and the designation date was changed July 1, 1982.

## 1. Scope

1.1 This practice covers procedures for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in the beltline of light-water cooled nuclear power reactor vessels. This practice includes guidelines for designing a minimum surveillance program, selecting materials, and evaluating test results.

1.2 This practice was developed for all light-water cooled nuclear power reactor vessels for which the predicted maximum neutron fluence ($E > 1$ MeV) at the end of the design lifetime exceeds $1 \times 10^{21}$ n/m$^2$ ($1 \times 10^{17}$ n/cm$^2$) at the inside surface of the reactor vessel.

## 2. Applicable Documents

2.1 *ASTM Standards:*

A 370 Methods and Definitions for Mechanical Testing of Steel Products[2]

E 8 Methods of Tension Testing of Metallic Materials[3]

E 21 Recommended Practice for Elevated Temperature Tension Tests of Metallic Materials[3]

E 23 Methods for Notched Bar Impact Testing of Metallic Materials[3]

E 208 Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels[3]

E 482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance[4]

E 560 Recommended Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results[4]

2.2 *American Society of Mechanical Engineers Standard*: Boiler and Pressure Vessel Code, Sections III and XI[5]

## 3. Significance and Use

3.1 Predictions of neutron radiation effects on pressure vessel steels are considered in the design of light-water cooled nuclear power reactors. Changes in system operating parameters are made throughout the service life of the reactor vessel to account for radiation effects. Because of the variability in the behavior of reactor vessel steels, a surveillance program is warranted to monitor changes in the properties of actual vessel materials caused by long-term exposure to the neutron radiation and temperature environment of the given reactor vessel. This practice describes the criteria that should be considered in planning and implementing surveillance test programs and points out precautions that should be taken to ensure that: (*1*) capsule exposures can be related to beltline exposures, (*2*) materials selected for the surveillance program are samples of those materials most likely to limit the operation of the reactor vessel, and (*3*) the tests yield results useful for the evaluation of radiation effects on the reactor vessel.

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[1] This practice is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications.

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[2] *Annual Book of ASTM Standards*, Vol 01.04.

[3] *Annual Book of ASTM Standards*, Vol 03.01.

[4] *Annual Book of ASTM Standards*, Vol 12.02.

[5] Available from the American Society of Automotive Engineers, 345 E. 47th St., New York, N. Y. 10017.

3.2 The design of a surveillance program for a given reactor vessel must consider the existing body of data on similar materials in addition to the specific materials used for that reactor vessel. The amount of such data and the similarity of exposure conditions and material characteristics will determine their applicability for predicting the radiation effects. As a large amount of pertinent data becomes available it may be possible to reduce the surveillance effort for selected reactors by integrating their surveillance programs.

## 4. Definitions

4.1 *adjusted reference temperature*—the reference temperature adjusted for irradiation effects by adding to $RT_{NDT}$ the transition temperature shift (see 4.15).

4.2 *base metal (parent material)*—as-fabricated plate material or forging material other than a weldment or its corresponding heat-affected-zone (HAZ).

4.3 *beltline*—the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core, and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material.

4.4 *EOL*—end-of-life; the design lifetime in terms of years; effective full power years; or neutron fluence.

4.5 *index temperature*—that temperature corresponding to a predetermined level of absorbed energy, lateral expansion, or fracture appearance obtained from the average (best fit) Charpy transition curve.

4.6 *fraction strength*—in a tensile test, the load at fracture divided by the initial cross-sectional area of the test specimen.

4.7 *fracture stress*—in a tensile test, the load at fracture divided by the cross-sectional area of the test specimen at time of fracture.

4.8 *heat-affected-zone* (HAZ)—plate material or forging material extending outward from, but not including, the weld fusion zone in which the microstructure of the base metal has been altered by the heat of the welding process.

4.9 *lead factor*—the ratio of the neutron flux density at the location of the specimens in a surveillance capsule to the neutron flux density at the reactor pressure vessel inside surface at the peak fluence location.

4.10 *neutron fluence*—the time integrated neutron flux density, expressed in neutrons per square metre or neutrons per square centimetre.

4.11 *neutron flux density*—a measure of the intensity of neutron radiation within a given range of neutron energies; the product of the neutron density and velocity, measured in neutrons per square metre-second or neutrons per square centimetre-second.

4.12 *neutron spectrum*—the distribution of neutrons by energy levels impinging on a surface, which can be calculated based on analysis of multiple neutron dosimeter measurements, on the assumption of a fission spectrum, or from a calculation of the neutron energy distribution.

4.13 *nil-ductility transition temperature* ($T_{NDT}$)—the maximum temperature at which a standard drop weight specimen breaks when tested in accordance with Method E 208.

4.14 *reference temperature* ($RT_{NDT}$)—See subarticle NB-2300 of the ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components."

4.15 *transition temperature shift* ($\Delta RT_{NDT}$) *or adjustment of reference temperature*—the difference in the 41-J (30-ft·lbf) index temperatures from the average Charpy curves measured before and after irradiation.

4.16 *transition region*—the region on the transition temperature curve in which toughness increases rapidly with rising temperature. In terms of fracture appearance, it is characterized by a rapid change from a primarily cleavage (crystalline) fracture mode to primarily shear (fibrous) fracture mode.

4.17 *Charpy transition curve*—a graphic presentation of Charpy data, including absorbed energy, lateral expansion, and fracture appearance, extending over a range including the lower shelf energy (< 5 % shear), transition region, and the upper shelf energy (> 95 % shear).

4.18 *upper shelf energy level*—the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy.

## 5. Test Materials

5.1 *Materials Selection*:

5.1.1 Surveillance test materials shall be prepared from samples taken from the actual materials used in fabricating the beltline of the reactor vessel. These surveillance test materials shall include one heat of the base metal, one butt weld, and one weld heat-affected-zone (HAZ). The base metal, weld metal, and HAZ (Note 1) materials included in the program shall be those predicted to be most limiting, with regard to setting pressure-temperature limits, for operation of the reactor to compensate for radiation effects during its lifetime (Note 2). The beltline materials shall be evaluated on the basis of initial reference temperature ($RT_{NDT}$), the predicted changes in the initial properties as a function of chemical composition (for example, copper (Cu) and phosphorus (P)) (Note 3), and the neutron fluence during reactor operation.

NOTE 1—The base metal for the weld heat-affected-zone (HAZ) to be monitored shall correspond to one of the base metals selected for the surveillance program.

NOTE 2—The data used for the selection of surveillance test materials shall be that obtained in accordance with ASME Code Section III requirements.

NOTE 3—Other residual/alloy elements such as Ni, Si, Mn, Mo, Cr, C, S, and V may contribute to overall radiation behavior of ferritic materials.

5.1.2 The base metal and the weld with the highest adjusted reference temperature at end-of-life shall be selected for the surveillance program. If the Charpy upper shelf energy of any of the beltline materials is predicted to drop to a marginal level (currently considered to be 68 J (50 ft·lbf) at the quarter thickness (¼ T) location) during the operating lifetime of the vessel, provisions shall be made to also include that material in the surveillance program, preferably in the form of fracture toughness specimens. These additional specimens may be substituted in part for specimens of the material least likely to be limiting.

5.1.3 The adjusted reference temperature of the materials in the reactor vessel beltline shall be determined by adding the appropriate values of transition temperature shift to the reference temperature of the unirradiated material. The transition temperature shift and Charpy upper shelf energy drop can be determined from relationships of fluence and chemical composition.

5.4 *Material Sampling*—A minimum test program shall consist of the material selected in 5.1, taken from the following locations: (*1*) base metal from one plate or forging used in the beltline, (*2*) weld metal made with the same heat of weld wire and lot of flux and by the same welding practice as that used for the selected beltline weld, and (*3*) the heat-affected-zone associated with the base metal noted above.

5.5 *Archive Materials*—Representative test stock to fill at least two additional capsules with test specimens of the base metal, weld, and heat-affected-zone materials used in the program shall be retained with full documentation and identification. It is recommended that this test stock be in the form of full-thickness sections of the original materials (plates, forgings, and welds).

5.6 *Fabrication History*—The fabrication history (austenitizing, quench and tempering, and post-weld heat treatment) of the test materials shall be fully representative of the fabrication history of the materials in the beltline of the reactor vessel and shall be recorded.

5.7 *Chemical Analysis Requirements*—The chemical analysis required by the appropriate product specifications for the surveillance test materials (base metal and as-deposited weld metal) shall be recorded and shall include phosphorus (P), sulfur (S), copper (Cu), vanadium (V), and nickel (Ni), as well as all other alloying and residual elements commonly analyzed for in low-alloy steel products. The product analysis shall be verified by analyzing a minimum of three test specimens randomly selected from both the base metal and the as-deposited weld metal.

## 6. Test Specimens

6.1 *Type of Specimens*—Charpy V-notch impact specimens corresponding to the Type A specimen described in Methods A 370 and E 23 shall be used. The gage section of irradiated and unirradiated tension specimens shall be of the same size and shape. Tension specimens of the type, size, and shape described in Methods A 370 and E 8 are recommended. Additional fracture toughness test specimens shall be employed to supplement the information from the Charpy V-notch specimens if the surveillance materials are predicted to exhibit marginal properties.

6.2 *Specimen Orientation and Location*—Tension and Charpy specimens representing the base metal and the weld heat-affected-zone shall be removed from about the quarter-thickness (¼ T) locations. Material from the mid-thickness of the plates shall not be used for test specimens. Specimens representing weld metal may be removed at all locations throughout the thickness with the exception of locations within 12.7 mm (½ in.) of the root or surfaces of the welds. The tension and Charpy specimens from base metal shall be oriented so that the major axis of the specimen is parallel to the surface and normal to the principal rolling direction for plates, or normal to the major working direction for forgings as described in Section III of the ASME Code. The axis of the notch of the Charpy specimen for base metal and weld metal shall be oriented perpendicular to the surface of the material; for the HAZ specimens, the axis of the notch shall be as close to perpendicular to the surface as possible so long as the entire length of the notch is located within the HAZ. The recommended orientation of the weld metal and HAZ specimens is shown in Fig. 1. Weld metal tension specimens may be oriented in the same direction as the Charpy specimens provided that the gage length consists entirely of weld metal. The weldment shall be etched to define the weld heat-affected-zone. The notch roots in the HAZ Charpy specimens shall be at a standard distance of approximately 0.8 mm (1/32 in.) from the weld fusion line. The orientation of the HAZ samples with respect to the major working direction of the parent material shall be recorded.

6.3 *Quantities of specimens*:

6.3.1 *Unirradiated Baseline Specimens*—It is recommended that 18 Charpy specimens be provided, of which a minimum of 15 specimens shall be tested to establish a full transition temperature curve for each material (base metal, HAZ, weld metal). The three remaining Charpy specimens should be reserved to provide supplemental data in instances such as excessive data scatter. At least three tension test specimens shall be provided to establish the unirradiated tensile properties for base metal and weld metal.

6.3.2 *Irradiated Specimens*—The minimum number of test specimens for each irradiation exposure set (capsule) shall be as follows:

| Material | Charpy | Tension |
|---|---|---|
| Base metal | 12 | 3 |
| Weld metal | 12 | 3 |
| HAZ | 12 | — |

It is suggested that a greater quantity of the above specimens be included in the irradiation capsules whenever possible.

## 7. Irradiation Requirements

7.1 *Encapsulation of Specimens*—Specimens should be maintained in an inert environment within a corrosion-resistant capsule to prevent deterioration of the surface of the specimens during radiation exposure. Care should be exercised in the design of the capsule to ensure that the temperature history of the specimens duplicates, as closely as possible, the temperature experienced by the reactor vessel. Surveillance capsules should be sufficiently rigid to prevent mechanical damage to the specimens and monitors during irradiation. The design of the capsule and capsule attachments shall also permit insertion of replacement capsules into the reactor vessel if required at a later time in the lifetime of the vessel. The design of the capsule holder and the means of attachment shall (*1*) preclude structural material degradation by the attachment welds, (*2*) avoid interference with inservice inspection required by ASME Code Section XI, and (*3*) ensure the integrity of the capsule holder during the service life of the reactor vessel.

7.2 *Location of Capsules*:

7.2.1 *Vessel Wall Capsules (Required)*—Surveillance capsules shall be located within the reactor vessel so that the specimen irradiation history duplicates as closely as possible, within the physical constraints of the system, neutron spectrum, temperature history, and maximum neutron fluence experienced by the reactor vessel. It is recommended that the surveillance capsule lead factors (the ratio of the instantaneous neutron flux density at the specimen location to the maximum calculated neutron flux density at the inside surface of the reactor vessel wall) be in the range of one to three. This range of lead factors will minimize the calculational uncertainties in extrapolating the surveillance measurements from the specimens to the reactor vessel wall and maximize the ability of the program to monitor material property changes throughout the life of the reactor vessel.

7.2.2 *Accelerated Irradiation Capsules (Optional)*—Additional test specimens may be positioned at locations closer to the core than those described in 7.2.1 for accelerated irradiation.

7.3 *Neutron Dosimeters:*

7.3.1 *Selection of Neutron Dosimeters*—Neutron dosimeters for the surveillance capsules shall be selected according to Guide E 482. The group of monitors selected shall be capable of providing fast neutron fluence, fast neutron spectrum, and thermal neutron flux density information. Dosimeters shall be included in every capsule.

7.3.2 *Location of Neutron Dosimeters*—Dosimeters shall be located within the vessel wall capsules (7.2.1) and the accelerated capsules (7.2.2) if used.

7.3.3 Separate dosimeter capsules should also be used to monitor radiation conditions independent of the specimen capsules if it is expected that the withdrawal schedule will otherwise result in saturation of the dosimeter activities.

7.4 *Correlation Monitors (Optional):*

7.4.1 *Selection of Correlation Monitor Materials*—Correlation monitors[6] have been found to be useful as an independent check on the measurement of irradiation conditions for the surveillance materials. Correlation monitor materials should be well characterized in terms of irradiation behavior (transition temperature shift). The magnitude of the transition temperature shift for this material should be measureable for the selected exposures.

7.5 *Temperature Monitors:*

7.5.1 *Selection of Temperature Monitors*—Major differences between specimen irradiation temperature and design temperature, occuring as a result of capsule design features, variation in reactor coolant temperature, or both, can affect the extent of radiation induced property changes in the surveillance materials. Since it is not practical to instrument the surveillance capsules, low melting point elements or eutectic alloys are used instead as monitors to detect significant variations in exposure temperature. These monitors are used in surveillance programs to provide evidence of the maximum exposure temperature of the specimens. The monitor materials should be selected to indicate unforeseen capsule temperatures.

7.5.2 *Location of Temperature Monitors*—One set of temperature monitors shall be located within the capsule where the specimen temperature is predicted to be the maximum. Additional sets of temperature monitors may be placed at other locations within the capsule to characterize the temperature profile.

7.6 *Number of Surveillance Capsules and Withdrawal Schedule:*

7.6.1 *Number of Capsules*—A sufficient number of surveillance capsules shall be provided to monitor the effects of neutron irradiation on the reactor vessel throughout its operating lifetime. The basis for the number of capsules to be installed at beginning of life is the predicted transition temperature shift, as shown in Table 1. The decrease in the upper shelf energy may also be a factor (see 5.1, 5.2, and 5.3). Additional capsules may be needed to monitor the effect of a major core change or annealing of the vessel, or to provide supplemental toughness data for evaluating a flaw in the beltline. It is recommended that full-thickness sections of material be kept instead of loaded capsules, because the preferred type and size of test specimen may change in the intervening years. The archive material required in 5.5 is to be used for the additional capsules.

7.6.2 *Withdrawal Schedule*—The capsule withdrawal schedule should permit monitoring of long-time effects which are difficult to achieve in test reactors. Table 1 lists the recommended number of capsules and the withdrawal schedule for three ranges of predicted transition temperature shift. The withdrawal schedule is in terms of effective full-power years (EFPY) of the vessel with a design life of 32 EFPY. Other factors that must be considered in establishing the withdrawal schedule are presented in Table 1. The first capsule is scheduled for withdrawal early in the vessel life to verify the initial predictions of the surveillance material response to the actual radiation environment. It is removed when the predicted shift exceeds the expected scatter by sufficient margin to be measureable. Normally, the capsule with the highest lead factor is withdrawn first. Early withdrawal will permit verification of the adequacy and conservatism of the reactor vessel pressure/temperature operational limits. The withdrawal schedule of the final two capsules is adjusted by the lead factor so the exposure of the second to last capsule does not exceed the peak end-of-life (EOL) fluence on the inside surface of the vessel, and so the exposure of the final capsule does not exceed twice the EOL vessel inside surface peak fluence. The decision on when to test specimens from the final capsule need not be made until the results from the preceding capsules are known.

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[6] Information regarding the availability of correlation monitors can be obtained from ASTM Committee E-10. See also ASTM DS54, July 1974.

7.6.3 *Implementation of Table 1:*

7.6.3.1 Estimate the peak vessel inside surface fluence at EOL and the corresponding transition temperature shift. This identifies the number of capsules required.

7.6.3.2 Estimate the lead factor for each surveillance capsule relative to the peak beltline fluence.

7.6.3.3 Calculate the number of EFPY for the capsule to reach the peak vessel EOL fluence at the inside surface and ¼ T locations. These are used to establish the withdrawal schedule for all but the first capsule.

7.6.3.4 Schedule the capsule withdrawals at the nearest vessel refueling date.

## 8. Measurement of Radiation Exposure Conditions

8.1 *Temperature Environment*—The maximum exposure temperature of the surveillance capsule materials shall be determined. If a discrepancy (> 14°C or 25°F) occurs between the observed and the expected capsule exposure temperatures, an analysis of the operating conditions shall be conducted to determine the magnitude and duration of these differences.

8.2 *Neutron Irradiation Environment:*

8.2.1 The neutron flux density, neutron energy spectrum, and neutron fluence of the surveillance specimens and the corresponding maximum values for the reactor vessel shall be determined in accordance with the guidelines in Guide E 482 and Recommended Practice E 560.

8.2.2 The specific method of determination shall be documented.

8.2.3 Neutron flux density and fluence values ($E > 0.1$ and 1 MeV) shall be determined and recorded using both a calculated spectrum and an assumed fission spectrum.

## 9. Measurement of Mechanical Properties

9.1 *Tension Tests:*

9.1.1 *Method*—Tension testing shall be conducted in accordance with Method E 8 and Recommended Practice E 21.

9.1.2 *Test Temperature:*

9.1.2.1 *Unirradiated*—The test temperatures for each material shall include room temperature, service temperature, and one intermediate temperature to define the strength *versus* temperature relationship.

9.1.2.2 *Irradiated*—One specimen from each material shall be tested at a temperature in the vicinity of the upper end of the Charpy energy transition region. The remaining specimens from each material shall be tested at the service temperature and the midtransition temperature.

9.1.3 *Measurements*—For both unirradiated and irradiated materials, determine yield strength, tensile strength, fracture load, fracture strength, fracture stress, total and uniform elongation, and reduction of area.

9.2 *Charpy Tests:*

9.2.1 *Method*—Charpy tests shall be conducted in accordance with Methods E 23 and A 370.

9.2.2 *Test Temperature:*

9.2.2.1 *Unirradiated*—Test temperatures for each material shall be selected to establish a full transition temperature curve. One specimen per test temperature may be used to define the overall shape of the curve. Additional tests should be performed in the region where the measurements described in 9.2.3 are made.

9.2.2.2 *Irradiated*—Specimens for each material will be tested at temperatures selected to define the full energy transition curve. Particular emphasis should be placed on defining the 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperatures and the upper shelf energy.

9.2.3 *Measurements*—For each test specimen, measure the impact energy, lateral expansion, and percent shear fracture appearance. From the unirradiated and irradiated transition temperature curves determine the 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperatures and the upper shelf energy. The index temperatures and the upper shelf energy shall be determined from the average curves.

9.2.3.1 Obtain from the material qualification test report the initial reference temperature ($RT_{NDT}$) as defined in the ASME Code, Section III, Subarticle NB 2300 for unirradiated materials.

9.3 *Hardness Tests (Optional)*—Hardness tests may be performed on unirradiated and irradiated Charpy specimens. The measurements shall be taken in areas away from the fracture zone or the edges of the specimens. The tests shall be conducted in accordance with Methods A 370.

9.4 *Supplemental Tests (Optional)*—If supplemental fracture toughness tests are conducted (in addition to tests conducted on tension and Charpy specimens as described in 6.1) the test procedures shall be documented.

9.5 *Calibration of Equipment*—Procedures shall be employed assuring that tools, gages, recording instruments, and other measuring and testing devices are calibrated and properly adjusted periodically to maintain accuracy within necessary limits.[7] Whenever possible calibration shall be conducted with standards traceable to the National Bureau of Standards. Calibration status shall be maintained in records traceable to the equipment.

## 10. Determination of Irradiation Effects

10.1 *Tension Test Data:*

10.1.1 Determine the amount of radiation strengthening by comparing unirradiated test results with irradiated test results at the temperatures specified in 9.1.2.

10.1.2 The tensile strength data can be verified using the results from the hardness test (optional) described in 9.3.

10.2 *Charpy Test Data:*

10.2.1 Determine the radiation induced transition temperature shifts by measuring the difference in the 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperatures before and after irradiation. The index temperatures shall be obtained from the average curves.

10.2.2 Determine the adjusted reference temperature by adding the shift corresponding to the 41-J (30-ft·lbf) index determined in 10.2.1 to the initial reference temperature obtained in 9.2.3.1.

10.2.3 Determine the radiation induced change in the upper shelf energy (USE) from measurements made before and after irradiation using average value curves.

10.2.4 *(Optional)*—Determine the radiation induced change in temperature corresponding to 50 % of the upper shelf energy before and after irradiation from average value curves.

10.3 *Supplemental Test Data (Optional)*—If additional, supplemental tests are performed (9.4), the data shall be recorded to supplement the information from the tensile and Charpy tests.

10.4 *Retention of Test Specimens*—It is recommended that all broken test specimens be retained until released by the owner in the event that additional analyses are required to explain anomalous results.

## 11. Report

11.1 The following information shall be provided. This report shall consist of the following elements. Where applicable, both SI units and conventional units shall be reported.

11.2 *Surveillance Program Description*—Description of the reactor vessel including the following:

11.2.1 Location of the surveillance capsules with respect to the reactor vessel, reactor vessel internals, and the reactor core.

11.2.2 Location in the vessel of the plates or forgings and the welds.

11.2.3 Location(s) of the peak vessel fluence.

11.2.4 Lead factors between the specimen fluence and the peak vessel fluence at the I.D. and the ¼ T locations.

11.2.5 *Surveillance Material Selection:*

11.2.5.1 Description of all beltline materials including chemical analysis, fabrication history, Charpy data, tensile data, drop-weight data and initial $RT_{NDT}$.

11.2.5.2 Describe the basis for selection of surveillance materials.

11.3 *Surveillance Material Characterization: tion*:

11.3.1 Description of the surveillance material including fabrication history, material source (heat or lot), and any differences between the surveillance material history and that of the reactor vessel material history.

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[7] Standardized specimens for certification of Charpy impact machines are available from the Army Materials and Mechanics Research Center, Watertown, Mass. 02172, Attn: DRXMR-MQ.

11.3.2 Location and orientation of the test specimens in the parent material.

11.3.3 *Test Specimen Design*:

11.3.3.1 Description of the test specimens (tension, Charpy, and any other types of specimens used), neutron dosimeters, and temperature monitors.

11.3.3.2 Certification of calibration of all equipment and instruments used in conducting the tests.

11.4 *Test Results*:

11.4.1 *Tension Tests*:

11.4.1.1 Trade name and model of the testing machine, gripping devices, extensometer, and recording devices used in the test.

11.4.1.2 Speed of testing and method of measuring the controlling testing speed.

11.4.1.3 Complete stress-strain curve (if a group of specimens exhibits similar stress-strain curves, a typical curve may be reported for the group).

11.4.1.4 Test data from each specimen as follows:

(*1*) Test temperature;

(*2*) Yield strength or yield point and method of measurement;

(*3*) Tensile strength;

(*4*) Fracture load, fracture strength, and fracture stress;

(*5*) Uniform elongation and method of measurement;

(*6*) Total elongation;

(*7*) Reduction of area; and

(*8*) Specimen identification.

11.4.2 *Charpy Tests*:

11.4.2.1 Trade name and model of the testing machine, available hammer energy capacity and striking velocity, temperature conditioning and measuring devices, and a description of the procedure used in the inspection and calibration of the testing machine.

11.4.2.2 Test data from each specimen as follows:

(*1*) Temperature of test;

(*2*) Energy absorbed by the specimen in breaking, reported in joules (and foot-pound-force);

(*3*) Fracture appearance;

(*4*) Lateral expansion; and

(*5*) Specimen identification.

11.4.2.3 Test data for each material as follows:

(*1*) Charpy 41-J (30-ft·lbf), 68-J (50-ft·lbf), and 0.89-mm (35-mil) lateral expansion index temperature of unirradiated material and of each set of irradiated specimens, along with the corresponding temperature increases for these specimens;

(*2*) Upper shelf energy (USE) absorbed before and after irradiation;

(*3*) Initial reference temperature; and

(*4*) Adjusted reference temperature.

11.4.3 *Hardness Tests (Optional)*:

11.4.3.1 Trade name and model of the testing machine.

11.4.3.2 Hardness data.

11.4.4 *Other Fracture Toughness Tests*:

11.4.4.1 If additional tests are performed, the test data shall be reported together with the procedures used for conducting the tests and analysis of the data.

11.4.5 *Temperature and Neutron Radiation Environment Measurements*:

11.4.5.1 Temperature monitor results and an estimate of maximum capsule exposure temperature.

11.4.5.2 Neutron dosimeter measurements, analysis techniques, and calculated results including the following:

(*1*) Neutron flux density, neutron energy spectrum, and neutron fluence in terms of neutrons per square metre and neutrons per square centimetre (> 0.1 and 1 MeV) for the surveillance specimens using both calculated spectrum and assumed fission spectrum assumptions.

(*2*) Description of the methods used to verify the procedures including calibrations, cross sections, and other pertinent nuclear data.

11.5 *Application of Test Results*:

11.5.1 Extrapolation of the neutron flux and fluence results to the surface and ¼ T locations of the reactor vessel at the peak fluence location.

11.5.2 Comparison of fluence determined from the dosimetry analysis with original predicted values.

11.5.3 Extrapolation of fracture toughness properties to the surface and ¼ T locations of the reactor vessel at the peak fluence location.

11.6 *Deviations*—Deviations or anomalies in procedure from this practice shall be identified and described fully in the report.

**TABLE 1 Minimum Recommended Number of Surveillance Capsules and Their Withdrawal Schedule (Schedule in Terms of Effective Full-Power Years of the Reactor Vessel)**

| | Predicted Transition Temperature Shift at Vessel Inside Surface | | |
|---|---|---|---|
| | ≤ 56°C (≤ 100°F) | > 56°C (> 100°F)<br>≤ 111°C (≤ 200°F) | > 111°C (> 200°F) |
| Minimun Number of Capsules | 3 | 4 | 5 |
| Withdrawal Sequence: | | | |
| First | $6^A$ | $3^A$ | $1.5^A$ |
| Second | $15^B$ | $6^C$ | $3^D$ |
| Third | $\text{EOL}^E$ | $15^B$ | $6^C$ |
| Fourth | | $\text{EOL}^E$ | $15^B$ |
| Fifth | | | $\text{EOL}^E$ |

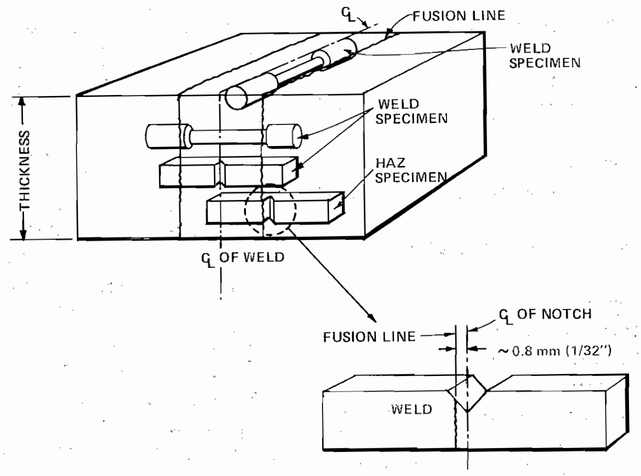
[A] Or at the time when the accumulated neutron fluence of the capsule exceeds $5 \times 10^{22}$ n/m$^2$ ($5 \times 10^{18}$ n/cm$^2$), or at the time when the highest predicted $\Delta RT_{NDT}$ of all encapsulated materials is approximately 28°C (50°F), whichever comes first.

[B] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

[C] Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel ¼ T location, whichever comes first.

[D] Or at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.

[E] Not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal.



(NOT DRAWN TO SCALE)

**FIG. 1 Location of Test Specimens Within Weld and Heat-Affected-Zone (HAZ) Test Material.**

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